NON-PUBLIC?: N
ACCESSION #: 9305180359
LICENSEE EVENT REPORT (LER)

FACILITY NAME: North Anna Power Station Unit 2 PAGE: 1 OF 4

DOCKET NUMBER: 05000339

TITLE: MANUAL REACTOR TRIP INITIATED DUE TO EXCESSIVE MAIN FEEDWATER REGULATING VALVE OSCILLATIONS
EVENT DATE: 04/24/93 LER #: 93-003-00 REPORT DATE: 05/12/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 2 POWER LEVEL: 071

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: G. E. Kane TELEPHONE: (703) 894-2101

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SJ COMPONENT: FCV MANUFACTURER: F130
B SJ SNB I207
B SJ H B209

REPORTABLE NPRDS: Y
N
N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On April 24, 1993, at 0530 hours with Unit 2 in Mode 1, 71 percent power, a manual reactor trip was initiated due to erratic feedwater flow indications to the "C" Steam Generator, abnormally high vibration alarms for the "B" Reactor Coolant Pump, and suspected water and steam leakage from the "C" Main Feedwater line. Emergency procedures were entered and immediate actions were performed. All Engineered Safety Feature system equipment responded as designed. At 0602 hours an "Alert" was declared, based on Shift Supervisor judgment, in accordance with Emergency Plan Implementing Procedures and the required notifications were made. A 1 hour report was made to the NRC at 0620 hours pursuant to 10CFR50.72 (a)

(ii) (3). At 0816 hours with the unit in a stable condition the "Alert" classification was down graded to a Notification Of Unusual Event (NOUE). The NOUE was terminated at 1045 hours on April 24, 1993. The event is reportable pursuant to 50.73 (a) (2) (iv).

The cause of the event has been attributed to the inherent instabilities with the design of the main feedwater regulating valve and positioner controls.

No significant safety consequences resulted from the main feedwater concerns because auxiliary feedwater was available and provided cooling to the steam generators throughout the event. Therefore, the health and safety of the public were not affected at any time during this event.

END OF ABSTRACT

1.0 Description of the Event

On April 24, 1993, at 0530 hours with Unit 2 in Mode 1, 71 percent power, a manual reactor trip was initiated due to erratic feedwater flow indications to the "C" Steam Generator (EIIS System AB, Component SG), abnormally high vibration alarms (EIIS System IV, Component VA) for the "B" Reactor Coolant Pump) (EIIS System AB, Component P), and suspected water and steam leakage from the "C" Main Feedwater line (EIIS System SJ). Emergency procedures were entered and immediate actions were performed. All Engineered Safety Feature system (EIIS System JE) equipment responded as designed. At 0602 hours an "Alert" was declared, based on Shift Supervisor judgment, in accordance with Emergency Plan Implementing Procedures and the required notifications were made. A 1 hour report was made to the NRC at 0620 hours pursuant to 10CFR50.72 (a) (ii) (3). At 0816 hours with the unit in a stable condition the "Alert" classification was down graded to a Notification Of Unusual Event (NOUE). The NOUE was terminated at 1045 hours on April 24, 1993. The event is reportable pursuant to 50 .73 (a) (2) (iv).

On April 24, 1993, at 0255 hours "C" Steam Generator (SG) feed flow began to oscillate. At 0320 hours "B" SG feed flow oscillated for approximately 10 minutes then reached steady state. At 0500 hours "B" SG feed flow began to oscillate again. At 0517 hours "C" feed flow oscillation worsened and both "B" and "C" Main Feed Regulating Valves (MFRV) (EIIS System SJ, Component FCV) were placed in manual. An operator was dispatched to the Mechanical Equipment Room and reported that the feed lines were moving and the "C" MFRV was cycling. Feedline noises and pipe movement worsened. The "B" Reactor Coolant Pump vibrations alarm was received and the "B" SG

Loose Parts alarm (EIIS System II, Component ALM) was received.

At 0530 hours after vibrations were felt and heard in the Control Room the Senior Reactor Operator ordered a manual reactor trip. The immediate actions of Emergency Procedure 2-E-0, Reactor Trip or Safety injection, were performed. Initially, Reactor Coolant System (RCS) (EIIS System AB) pressure decreased to approximately 2000 psig and temperature decreased to approximately 538 degrees F. All Engineered Safety Feature system equipment responded as designed. Upon transition to Emergency Procedure 2-ES-0.1, Reactor Trip Response, the secondary systems were secured. At this point the noises and vibrations were no longer heard or felt in the Control Room. At 0602 hours an "Alert" was declared, based on Shift Supe
visor judgment, in accordance with Emergency Plan Implementing Procedures. Notification of State and Local Governments occurred at 0609 hours. The NRC Emergency Operations Center was notified of the "Alert" at 0620 hours pursuant to 10CFR50.72 (a) (ii) (3). At 0816 hours with the unit in a stable condition the "Alert" classification was down graded to a Notification Of Unusual Event (NOUE). The NRC was notified at 1045 hours that the NOUE was terminated. At 1253 hours the unit entered Hot Shutdown with RCS temperature less than 350 degrees F. At 1735 hours the unit entered Cold Shutdown with RCS temperature less than 200 degrees F.

2.0 Significant Safety Consequences and Implications

No significant safety consequences resulted from the main feedwater concerns because auxiliary feedwater was available to provide cooling to the steam generators throughout the event. Therefore, the health and safety of the public were not affected at any time during this event.

3.0 Cause of the Event

The cause of the event has been attributed to the inherent instabilities with the design of the main feedwater regulating valve and positioner controls.

4.0 Immediate Corrective Actions

Following the manual reactor trip Emergency Procedure 2-E-0, Reactor Trip or Safety Injection, was entered and the immediate actions performed.

Additional Corrective Actions

Upon transition to Emergency Procedure 2-ES-0.1, Reactor Trip Response, the secondary systems were secured. The unit was brought to cold shutdown without incident.

The flow and pressure transmitters for the "C" main feedwater lines were satisfactorily calibrated and verified operable. The MFRVs and Main Feedwater Bypass Regulating Valves were visually inspected, stroked, calibrated, and verified operable. The feedwater piping and supports downstream of "C" MFRV up to and including the feed ring inside "C" SG were inspected. Adjustments were made to the MFRVs pneumatic boosters to reduce sensitivity and overshoot.

6.0 Actions to Prevent Recurrence

Guidance to mitigate oscillations was provided to all operations shifts. Early recognition of feedwater system oscillation during power ascension will allow appropriate operator actions to occur to preclude a similar event. Parameters that will be monitored include: feedwater pump suction pressure, feedwater header pressure, feedwater flow, and the individual pressure indications at the discharge of each MFRV.

Requirements are in place to ensure the reactor trip event is included in the Licensed Operator Requalification Program. Actions taken are sufficient to preclude recurrence.

7.0 Similar Events

None

8.0 Additional Information

Component failures resulting from the main feedwater line oscillations include: broken air line to the "B" Main Feedwater Regulating Valve, broken air line to the "B" Main Feedwater Bypass Regulating Valve, broken snubber extension rod on the "C" Main Feedwater Bypass line, broken spring hanger on "C" Feedwater line, and a broken yoke on "C" main feedwater bypass regulating valve.

The broken air lines to the "B" MFRV and "B" Main Feedwater Bypass Regulating Valve were repaired. The "C" Main Feedwater Bypass line snubber was replaced. Snubbers were also replaced on the "A" and "C" feedwater headers. The spring hanger on "C" Feedwater line was repaired and the yoke on "C" main feedwater bypass regulating valve was replaced.

Unit 1 was in Mode 1, 100 percent power, and was not affected by the event.

The Unit 1 MFRVs were modified during the recent refueling outage with a pneumatic open and close operator. This modification was performed to eliminate oscillations of the MFRVs. The Unit 1 MFRV are of a different design than Unit 2 and experience to date indicates they do not have similar instabilities.

ATTACHMENT 1 TO 9305180359 PAGE 1 OF 1

Vepco VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION

P. O. BOX 402

MINERAL, VIRGINIA 23117

10 CFR 50.73

May 12, 1993

U. S. Nuclear Regulatory Commission NAPS:MPW
Attention: Document Control Desk Docket No. 50-339
Washington, D.C. 20555 License No. NPF-7

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 2.

Report No. 50-339/93-003-00

This Report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Corporate Management Safety Review Committee for its review.

Very Truly Yours,

G. E. Kane
Station Manager

Enclosure:

cc: U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. D. R. Taylor
NRC Resident Inspector
North Anna Power Station

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